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Abstract

ANALYSIS OF INTEGRAL DATA FOR FEW GROUP PARAMETER EVALUATION OF FAST REACTORS

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The most critical problem arising in the evaluation of fast reactor performance is the lack of reliable data for many important nuclear processes. Those available are derived from scattered differential measurements or from some specific nuclear model. Generally, cross-section sets obtained in different laboratories differ considerably from one another. This situation is particularly serious for fast systems employing the U-233-Th cycle. The difficulty inherent in the differential measurements of these parameters suggests the exploitation of some integral data obtainable in a critical facility in order to arrive at a final few-group set with a validity within a certain range of systems.

In the first part of this paper an analysis is made on the applicability of a single few-group cross-section set to a broad range of reactors. This validity is, of course, dependent on the number of groups and for this reason a sixteen group set has been chosen, which represents a sufficiently standard few-group set and is compatible with the maximum number of groups permissible for many one or two dimension diffusion or transport codes. A series of critical facilities are then surveyed and compared with regard to the representation of the spectrum of the reference system in the test region and precision of the measurements. Thermal drivers are compared with fast drivers either for zero flux gradient measurements or for transport measurements which are assumed to be of paramount importance for fast reactor performance evaluation. In the thermal driven facilities, the flux gradient can be achieved either through an internal exponential experiment or by means of transversally finite slab geometry. Finally, a method is described for the reduction of the integral data into few-group parameters starting from an assumed best value set.